NON-PUBLIC?: N
ACCESSION #: 8910030090
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit 1 PAGE: 1 OF 09

DOCKET NUMBER: 05000413

TITLE: Manual Reactor Trip Due to Failure of a Gasket on The Main Feedwater Valve Positioner Control Air Manifold
EVENT DATE: 8/23/89 LER #: 89-022-00 REPORT DATE: 09/21/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: R. M. Glover, Compliance Manager TELEPHONE: (803)831-3236

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SJ COMPONENT: HHF MANUFACTURER: M430
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On August 24, 1989, Unit 1 was in Mode 1, Power Operation, at 100% power. At 1930 hours, 1CF28, Steam Generator (S/G) 1A Main Feedwater (CF) Control Valve, began closing, causing a Steam Generator 1A level deviation alarm. The Control Room Operators (CROs) responded by opening 1CF30, S/G 1A CF Control Bypass Valve, and dispatching Operations personnel and Instrumentation and Electrical (IAE) personnel to investigate the problem. The CRO began power reduction to compensate for the reduced CF flow to S/G 1A. At 1944 hours, S/G 1A level began dropping rapidly. The CRO then initiated a manual Reactor trip and entered Emergency Procedure EP/1/A/5000/01, Reactor Trip or Safety Injection. By 2110 hours, S/G 1A was returned to the normal operating level. 1CF28 malfunctioned due to a failed gasket in the positioner control air manifold. The gasket appeared to have failed due to improper design and/or installation deficiency. The positioner was replaced on 1CF28 and on the remaining three Unit 1 Control Valves before Unit 1 returned to Mode 1, Power Operation, at 0753 hours on August 25, 1989. The CF Control Valve positioners are being inspected weekly for air

leaks, and an improved gasket design has been implemented on Unit 1.

END OF ABSTRACT

BACKGROUND

The Main Feedwater EIIS:SJ! (CF) System supplies feedwater to the four Steam Generators EIIS:HX! (S/Gs) at the temperature, pressure, and flow required to maintain proper S/G water levels commensurate with Reactor power output and Turbine (EIIS:TRB! steam requirements. The CF System contains two 50% capacity variable speed Turbine Driven CF Pumps EIIS:P! (CFPTs). The CF pump speeds may be manually or automatically controlled. The CF pumps discharge through two stages of high pressure CF heaters EIIS:HTR!. The feedwater then divides into four CF lines, each supplying one of the four S/Gs. Each of the four CF lines contains a CF Control valve EIIS:V!, a CF Control Bypass valve, two CF Check valves, and a CF Isolation valve. The CF Isolation valves function to terminate CF flow in either direction following a CF Isolation signal and also function to prevent or allow admission of feedwater to the S/Gs CF nozzles during various modes of operation. The CF Control Bypass valves normally are utilized to control CF flow up to approximately 15% load, and the CF Control valves are utilized to control CF flow from approximately 15% to 100% load.

The CF Control valves are normally automatically controlled by the S/G Level Control System to maintain proper S/G levels. Each CF Control valve has an electrical to pneumatic (E/P) control converter EIIS:XB! which converts the 4 to 20 milliamp signal from the control system to supply a 3 to 15 psi pneumatic signal to the control valve positioner. The positioner is mechanically linked to the valve stem and regulates the air pressure to the CF control valve actuator EIIS:XCV! to match the CF cont
ol valve position with the demand signal from the E/P converter.

EVENT DESCRIPTION

On August 24, 1989, Unit 1 was in Mode 1, Power Operation, at 100% power. At 1930 hours, the Control Room Operators (CROs) received a S/G 1A Level Deviation Annunciator EIIS:ANN! and noted a full open demand signal for 1CF28, S/G 1A Main Feedwater (CF) Control valve. The CROs responded by taking manual control of 1CF28 and opening 1CF30, S/G 1A CF Control Bypass valve. The Operations Unit Supervisor, a Nuclear Operations Technician and Instrumentation and Electrical (IAE) personnel were dispatched to locally investigate the problem on 1CF28 and found air leaking at the positioner manifold. The IAE Technician attempted to block the air leak by applying finger pressure against the leak. However, S/G 1A level continued to decrease. At 1942 hours, the CROs had fully opened 1CF30 and at 1943 hours, turbine load had been reduced to approximately 92%. At 1944:19 hours, the CRO initiated a manual Reactor trip due to the rapidly decreasing level in S/G 1A. The turbine tripped on a Reactor trip, as expected. At 1944:20 hours, the S/G 1A Lo Lo Level Reactor Trip Signal was received, the Motor EIIS:MO! Driven Auxiliary Feedwater EIIS:BA! (CA) pumps autostarted, and the S/G Blowdown EIIS:WI! (BB) System and Nuclear Sampling EIIS:KN! (NM) System isolated, as expected. Feedwater Isolation occurred, as expected, due to the Reactor Trip concurrent with low Reactor Coolant temperature of 564 degrees F.

The post-trip cooldown was controlled by the automatic cycling of the bank 1, bank 2, and bank 3 Main Steam Bypass to Condenser EIIS:SO! (SB) valves. Six of the nine SB valves indicated open to control steam pressure, however 1SB24, Main Steam Bypass to Condenser Controller #24, and 1SB21, Main Steam Bypass to Condenser Controller #21, did not indicate open and 1SB27, Main Steam Bypass to Condenser Controller #27, was isolated for repair. The four S/G Power Operated Relief Valves (PORVs) also cycled to control steam pressure. At 1945:36 hours, the two operating Condensate Booster Pumps tripped due to a low flow trip signal. At 1946:21 hours, the CF Pump 1B tripped due to low suction flow apparently as a result of the Condensate Booster Pump trip. Condensate Booster Pump 1A automatically started at 1952 hours.

By 2007 hours, S/G 1A level had been returned to the normal operating level. Subsequently, the leaking positioner on 1CF28 was replaced per Work Request 51376 OPS. The Positioner manifold gaskets on the remaining Unit 1 CF Control valves were also replaced per Work Requests 51384 OPS, 51385 OPS, and 51386 OPS.

By 2035 hours, the CRO exited EP/1/A/5000/1A, Reactor Trip Response, and entered OP/1/A/6100/05, Unit Fast Recovery. By 2115 hours, the CRO reset the CF Isolation signal, realigned the NM and BB Systems, placed the CF Control Bypass valves in automatic, and secured the CA pumps.

After repair of the CF control valve positioners, Unit 1 entered Mode 2, Startup, at 0608 hours on August 25, 1989. At 0753 hours, Unit 1 entered Mode 1 and began power escalation to 100%.

On August 25, 1989, the four Unit 2 positioner gaskets were replaced and Permatex sealant was applied to the gasket surfaces per Work Requests

10668 IAE, 10669 IAE, 10670 IAE and 10671 IAE. At that time Unit 2 was below 15% power, not requiring operation of these control valves.

Examination of the failed positioner prior to disassembly of the manifold and diaphragm assembly (see Page 8) found approximately 1/4 inch of the gasket protruding from the edges of the mating surfaces. The torques required to loosen the two manifold screws were 25 in.lb and 35 in.lb. After disassembly, the gasket relaxed back to its original shape and was observed to be torn between the supply air port and the manifold screw opening (see Page 9). This gasket failure appears similar to the gasket failure that occurred on June 26, 1989, addressed in LER 413/89-017.

During the examination of the manifold and manifold gaskets, it was noted that the supply air opening in the gasket is much larger than the opening required by the size and relative location of the ports of the mating components. It was also noted that two manifold mounting screws in the parts list are specified as 5/16-18NC X 1 3/8 inches long round head screws. The screws from the leaking positioner were measured to be 1 3/16 inches to 1 5/16 inches long and appeared to engage 3 threads when tightened. Moore Products Company was subsequently contacted to determine: 1) if the gasket design had been modified since the

original gaskets had been ordered, 2) the reason for the large supply port opening, and 3) if there is a recommended torque for the manifold screws. The manufacturer indicated that the gasket had not been modified since 1977 and that the large supply port was due to a generic design to fit other manifolds. The manufacturer indicated that the manifold screws are factory torqued to 55 in.lb.

On September 1, 1989, Work Request 1366 MES was initiated to repair an air leak on the positioner gasket for 1CF46, S/G 1C CF Control valve. Due to observations from the previous gasket failures, Nuclear Station Modification (NSM) VN CE-2442 was issued to fabricate a modified manifold gasket by providing a smaller supply port opening, using longer manifold mounting screws, and using a thicker gasket material. The modified gaskets were installed on the Unit 1 CF Control valves on September 3, 1989. Permatex sealant was applied to the gasket and the two manifold screws were torqued to 55 in.lb. Unit 1 was subsequently returned to 100% power.

CONCLUSION

This Reactor trip was caused by the failure of a gasket on the air manifold for the 1CF28 valve positioner. The failure appears to be due to inadequate torque on the manifold mounting screws and/or marginal gasket design. Torque specifications were not provided by the manufacturer. The Moore Products positioners were installed in May of 1985 on Unit 1 per NSM CN-10346 to replace the original Bailey positioners which had been discontinued. A review of work request history does not reveal any problems with the gaskets on the Moore Products positioners prior to June 16, 1989, when 1CF28 was found to have developed a leak on the manifold gasket which was repaired on Work Request 1232 MES. This gasket was described as torn and brittle and was replaced with a new gasket. Ten days later, Unit 1 tripped due to a failed manifold gasket on the 1CF28 valve positioner, as addressed in LER 413/89-017. The LER attributed the gasket failure to gasket damage prior to or during installation due to a tear in the gasket between the manifold bolt opening and the supply air opening. Gouge marks were also present in the aluminum surface of the mating surface of the diaphragm assembly. The positioner assembly was replaced per Work Request 50192 OPS on June 26, 1989. The failure of the gasket on August 24, 1989, was similar except that no gouge marks were present on the mating surfaces. Since the manifold positioner gaskets gave satisfactory service for five years prior to the replacement of the gasket in 1CF28 and since air leaks have been a problem since the gaskets have been replaced, it is concluded that the high incidence of gasket leaks may be attributed to inadequate or unequal torque on the manifold screws. Marginal gasket design may have also contributed to the gasket failure since the larger than required supply air opening has as little as 1/4 inch of compression surface. NSM VN CE-2442 reduced the size of the gasket supply port opening from 5/8 inch to 1/4 inch which approximately doubled the minimum compression surface. Over pressure of the valve positioner supply air regulator for 1CF28 was also considered as two failures have occurred on 1CF28. However, the positioner supply is rated for 100 psi maximum which is greater than the normal instrument air EIIS:LP! and the pressure regulator to 1CF28 appears to be operating satisfactorily.

Gasket failures on Unit 1 have caused Reactor trips two times in the past twelve months. This is considered to be a recurring event.

After the Reactor trip on August 24, 1989, 1SB21 and 1SB24 did not indicate open. Performance personnel discussed this with Maintenance Engineering Services personnel and concluded that the valves actually opened. However, 1SB24 did not open on a subsequent Load Rejection on September 13, 1989. Therefore, Work Request 6743 SWR was issued to repair 1SB24.

The Condensate Booster pumps tripped unexpectedly on low flow approximately one minute after the Reactor trip and appeared to have caused the trip of CF Pump 1B. Work Request 7221 PRF has been issued to check the Condensate Booster Pump low flow pressure switches to determine the cause of the Condensate Booster Pumps low flow trip.

CORRECTIVE ACTION

IMMEDIATE

1) The CRO opened 1CF30 to provide additional CF flow.

2) The CRO manually tripped the Reactor when the S/G 1A levels began rapidly decreasing and entered EP/1/A/5000/01.

SUBSEQUENT

1) The CROs stabilized the Unit in Mode 3, Hot Standby, per EP/1/A/5000/1A, Reactor Trip Response.

2) The positioner on 1CF28 was replaced per Work Request 51376 OPS.

3) The positioner manifold gaskets in the remaining Unit 1 Control Valves were replaced per Work Requests 51384 OPS, 51385 OPS, and 51386 OPS.

4) The manifold gaskets on the Unit 2 CF Control Valve positioners were replaced using Permatex sealer on the gasket surfaces per Work Requests 10668 IAE, 10669 IAE, 10670 IAE and 10671 IAE.

5) Moore Products Company was contacted to determine the recommended manifold torque, the reason for the large supply air opening and if gasket modifications had been made.

6) NSM CEVN 2442 was issued to modify the manifold gasket by reducing the size of the supply air opening, using a thicker gasket material, and to specify longer manifold screws.

7) NSM CEVN 2442 was implemented on Unit 1 when power was reduced to repair 1CF46.

8) IAE is inspecting the CF Control Valve positioners weekly for air leaks.

PLANNED

1) IP/O/B/3820/30, Calibration and Maintenance Procedure for Moore Side Mounted Positioners, will be revised to include a 55 in.lb torque requirement for the positioner manifold screws and specify appropriate torques for other positioner components.

2) Condenser Steam Dump Valve 1SB24 will be repaired per Work Request 6743 SWR.

3) Unit 1 CM System low flow pressure switch setpoints will be checked per Work Request 7221 PRF.

4) The slower operation of the S/G 1B PORV will be investigated per Work Request 3984 SWR.

5) Minor discrepancies noted during the post-trip review will be identified on the Station tracking document.

SAFETY ANALYSIS

Due to the modulation of 1CF28, moderator feedback and steam load decrease reduced Reactor power from 100% to 95% in approximately 5.5 minutes. The Operator initiated a Unit runback by reducing turbine/generator load with the control rods in automatic. The manual Unit runback reduced Reactor power from 95% to 91%. A manual Reactor trip was initiated at 91% full power in anticipation of a S/G low-low level Reactor trip signal. The Reactor trip breakers opened within 46 milliseconds of the manual Reactor trip signal. All of the control rods fell to the bottom of the core, reducing power to decay heat level. Approximately 1 second after the manual Reactor trip, S/G 1A low-low level signal occurred, autostarting the motor driven CA pumps. Feedwater Isolation occurred due to Reactor trip with low Tave logic at the design setpoint of 564 degrees F.

Prior to Reactor trip, Reactor Coolant EIIS:AB! System cold leg temperature in Loops B, C, and D increased approximately 2 degrees F to 563 degrees F, and Loop A temperature increased approximately 6 degrees F to 568 degrees F. Reactor Coolant System hot leg temperature remained stable at 620 degrees F. Upon Reactor trip, Reactor Coolant System Tave decreased and stabilized at the no-load target of 557 degrees F. Pressurizer pressure fluctuated slightly around 2235 psig prior to Reactor trip, and upon Reactor trip decreased to a minimum value of 2010 psig. Pressurizer pressure increased to 2260 psig within 40 minutes post-trip, 25 psi from the no-load target of 2235 psig. Pressurizer spray automatically initiated to limit the pressure increase.

Pressurizer level decreased from approximately 62% to a minimum value of 24% upon Reactor trip,

and stabilized at 27% within 40 minutes post-trip, 2% from the no-load target of 25%. Steam pressure increased during modulation and closure of 1CF28 from 1010 psig to 1040 psig. Due to the increase in steam pressure prior to Reactor trip, steam pressure increased to a maximum value of 1140 psig upon Reactor trip. Steam pressure stabilized at approximately 1060 psig within 40 minutes post-trip, 30 psi from the no-load target of 1090 psig. S/Gs 1B, 1C, and 1D narrow range levels remained on-scale post-trip, and S/G 1A wide range level decreased to a minimum indicated value of 34% post-trip. Correction of this value for calibration condition variations yields an actual minimum wide range level of 43%. S/G narrow range levels stabilized at an average value of approximately 33% within 40 minutes post-trip.

The Unit runback prior to Reactor trip did not cause a significant enough deviation between Tave and Tref to open the steam dump valves. After Reactor trip, the steam dump to condenser valves and the S/G PORVs opened to limit steam pressure increase and remove heat. One steam dump valve in bank 1 was isolated, and one steam dump valve each in banks 2 and 3 failed to indicate open. As required by the Reactor Trip or Safety injection Emergency Procedure, CA flow was maintained greater than 450 gpm while S/G wide range level indication was less than 47%. Reactor coolant was 45 degrees F subcooled at the point of minimum pressurizer pressure. Adequate core decay heat removal was available and maintained at all times.

S/G 1B PORV, 1SV13, was slow in opening and closing in automatic control as compared to the other PORVs. The safety related function of the S/G PORVs is to assist in mitigating the consequences of a postulated S/G U-tube rupture concurrent with a loss of offsite power. In this scenario, automatic operation of the S/G PORVs is not a concern because the operator assumes manual control of the PORVs. 1SV13 achieved a fully open position, so the sluggish operation of the valve was related to setpoint and not the response to a demand to open. Constrictive time response problems are therefore not created for this accident scenario, and the safety related function of 1SV13 was not impaired.

Reactor Coolant System pressure and inventory control functioned as designed throughout this event. All safety related structures, systems and components were available during this event. Primary and secondary side no-load targets were approached post-trip, and the cooldown limits of 100 degrees F per hour for the Reactor Coolant System and 200 degrees

F per hour for the pressurizer were not exceeded. This event is fully bounded by the "Loss of Normal Feedwater Flow" transient as described in Section 15.2.7 of the Catawba FSAR. This event was less severe than the total loss of feedwater as assumed in the FSAR. Also, the FSAR assumes operability of only the S/G code safety valves. In this event, the S/G PORVs and the steam dump to condenser valves were additionally available to remove heat. The health and safety of the public were not affected by this event.

TEXT PAGE 8 OF 09

Figure entitled "Moore Products Positioner Schematic" omitted.

TEXT PAGE 9 OF 09

Figure entitled "Manifold Gasket" omitted.

ATTACHMENT 1 TO 8910030090 PAGE 1 OF 1

Duke Power Company (803) 831-3000
Catawba Nuclear Station
P.O. Box 256
Clover, S.C. 29710

DUKEPOWER

September 21, 1989

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/89-22

Gentlemen:

Attached is Licensee Event Report 413/89-22, concerning manual reactor trip due to failure of a gasket on the main feedwater valve positioner control air manifold.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Tony B. Owen
Station Manager

KEB\LER-NRC.TBO

xc: Mr. S. D. Ebneter American Nuclear Insurers
Regional Administrator, Region II c/o Dottie Sherman, ANI Library
U. S. Nuclear Regulator Commission The Exchange, Suite 245
101 Marietta Street,NW, Suite 2900 270 Farmington Avenue
Atlanta, GA 30323 Farmington, CT 06032

M & M Nuclear Consultants Mr. K. Jabbour
1221 Avenues of the Americas U. S. Nuclear Regulatory
New York, NY 10020 Commission
Office of Nuclear Reactor
Regulation
Washington, D.C. 20555
INPO Records Center
Suite 1500 Mr. W. T. Orders
1100 Circle 75 Parkway NRC Resident Inspector
Atlanta, GA 30339 Catawba Nuclear Station

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